10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
1022 Haley Drive
Mineral, Virginia 23117

April 24, 2015

Attention: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Serial No.: 15-152
NAPS: RAP
Docket No.: 50-338
License No.: NPF-4

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Power Station Unit 1.

Report No. 50-338/2015-001-00

This report has been reviewed by the Facility Safety Review Committee and will be forwarded to the Management Safety Review Committee for its review.

Sincerely,

Gerald T. Bischof
Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

NRC Senior Resident Inspector
North Anna Power Station

IE22
NRR

U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See Page 2 for required number of digits/characters for each block)

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4. TITLE

Automatic Reactor Trip Due to Low-Low Level on “B” Steam Generator

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	26	2015	2015	- 001	- 00	04	24	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 96

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
Gerald T. Bischof, Site Vice President	(540) 894-2101

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	FCV	W120	Y	A	BA	65	W290	Y

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On February 26, 2015, at 1511 hours with Unit 1 operating at 96 percent power in an end of cycle coastdown, an automatic reactor trip occurred. The initiating signal was a low-low level on “B” steam generator caused by the closure of the “B” main feedwater regulating valve (MFRV). Closure of the valve was due to a loss of power on the final driver card for “B” MFRV. At 1639 hours, a 4-hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72(b)(2)(iv)(B) for “an event causing actuation of the Reactor Protection System when the reactor is critical” and an 8-hour report was also made in accordance with 10 CFR 50.72(b)(3)(iv)(A) for “an event causing actuation of the Auxiliary Feedwater System (AFW).” Approximately 30 minutes after the reactor trip with AFW flow throttled, the Turbine Driven (TD) AFW pump, 1-FW-P-2, discharge relief valve (RV) lifted and discharged approximately 200 gallons per minute (gpm) to the ground. The RV opened as a result of the inability of the governor valve for the pump to travel an additional 3/16” in the closed direction. An engineering evaluation determined that the loss of emergency condensate storage tank inventory was a condition reportable per 10 CFR 50.73(a)(2)(i)(B) for an “any operation or condition which was prohibited by Technical Specifications.” The health and safety of the public were not affected by either event.

NARRATIVE

1.0 DESCRIPTION OF THE EVENT

On February 26, 2015, at 1511 hours with Unit 1 operating at 96 percent power in an end of cycle coastdown, an automatic reactor trip occurred. The initiating signal was a low-low level on the “B” steam generator (SG) (EIIS System AB, Component SG) caused by closure of the “B” main feedwater regulating valve (MFRV) (EIIS System SJ, Component FCV). This resulted in a reactor and turbine trip.

Following the reactor trip the Reactor Protection System (RPS) and all Engineered Safety Feature Actuation System (ESFAS) (EIIS System JE) equipment actuated as designed, including Alternate Mitigation System Actuation Circuitry (AMSAC) and the Auxiliary Feedwater (AFW) (EIIS System BA) pumps. The control room operators responded to the event in accordance with emergency procedure 1-E-0, Reactor Trip or Safety Injection. The operators then stabilized the plant using 1-ES-0.1, Reactor Trip Recovery.

At 1639 hours, a 4-hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72(b)(2)(iv)(B) for “an event causing actuation of the Reactor Protection System when the reactor is critical” and an 8-hour report was also made in accordance with 10 CFR 50.72(b)(3)(iv)(A) for “an event causing actuation of the Auxiliary Feedwater System.”

Approximately 30 minutes after the reactor trip with AFW flow throttled, the Turbine Driven (TD) AFW pump, 1-FW-P-2, (EIIS System BA, Component P) discharge relief valve (EIIS System BA, Component RV) lifted and discharged approximately 200 gallons per minute (gpm) to the ground. Troubleshooting determined that while the TD AFW pump was throttled back on its head curve via 1-FW-MOV-100D (EIIS System BA, Component 20) to near maximum recirculation, the relief valve opened as a result of the inability of the governor valve, 1-FW-GOV-2-VALVE (EIIS System BA, Component 65), which regulates steam flow to the TD AFW pump, to travel an additional 3/16” in the closed direction. This was due to improper installation of the governor valve during maintenance. The improper installation resulted in steam leakage past the governor valve to the TD AFW pump.

An engineering evaluation determined that with the loss of the approximately 200 gpm of Emergency Condensate Storage Tank (ECST) (EIIS System BA, Component TK) inventory, the ECST could not have met its mission time for certain accident conditions. However, there are multiple water sources via the Condensate header makeup capability to refill the ECST as well as the Fire Protection system, the Service Water system, and the Beyond Design Basis (BDB) connections which provide AFW so the safety significance of this issue is low. This condition is reportable in accordance with 10 CFR 50.73(a)(2)(i)(B) for an “any operation or condition which was prohibited by

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Technical Specifications."

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

No significant safety consequences resulted from this event because the RPS and ESFAS systems actuated as designed following the trip. The health and safety of the public were not affected by this event.

Given the extensive network of defense-in-depth capability, via the Condensate header makeup capability to refill the ECST and the Fire Protection system, the Service Water system, and the BDB connections to provide AFW, the safety significance regarding the previously inoperable condition of the ECST is considered low and did not present a threat to the health and safety of the public.

3.0 CAUSE

The direct cause of the "B" MFRV closure while in automatic and subsequent reactor trip was a failure of 1-FW-FCY-1488, final driver card for "B" MFRV, due to a loss of power. The root cause leading to the failure of 1-FW-FCY-1488 was contamination within a silicon controlled rectifier (SCR) on the circuit board power supply. The investigation found that the SCR prematurely failed due to embedded contaminates within the device package originating from the manufacturing process. The contaminates caused progressive thermal damage which led to the premature failure of the SCR. The failure was a short circuit condition which resulted in a loss of input power to the power supply.

The apparent cause for 1-FW-P-2 running at a pump speed outside of the acceptable range is the incorrect installation of the governor valve linkage in accordance with 0-MCM-0412-02. Procedure steps for correctly installing the governor valve linkage were performed without obtaining the correct outcome.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

The control room operators responded to the event in accordance with emergency procedure 1-E-0, Reactor Trip or Safety Injection. The operators then stabilized the plant using 1-ES-0.1, Reactor Trip Recovery. All safety systems responded appropriately. The unit was stabilized at no-load conditions, the Main Feedwater system was placed in service to all three SGs, and the AFW system was secured and returned to AUTO, with 1-FW-P-2 declared inoperable.

5.0 ADDITIONAL CORRECTIVE ACTIONS

Design change DC NA-15-00031 was implemented to remove the SCRs on the Unit 1

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7300 system NCD and NMD circuit card assemblies that had been identified as single point vulnerabilities.

The governor valve for 1-FW-P-2 was repaired and the pump was declared operable at 1609 hours on February 28, 2015. Additionally, 0-MCM-0412-02 is being improved to help ensure the correct outcome is obtained in the future. Also, the post-maintenance testing is being enhanced to ensure the governor valve operates correctly for all pump operating conditions.

6.0 ACTIONS TO PREVENT RECURRENCE

A similar design change to that of DC NA-15-00031 is being prepared for Unit 2 and will be implemented during the next outage of sufficient duration. Additional design changes are being created for each unit to eliminate these single point vulnerability cards on the MFRVs, as well as on other control systems. The design changes for the MFRVs were already scheduled for the respective 2016 refueling outages.

7.0 SIMILAR EVENTS

LER N1-07-001-00 dated 02/27/07, documents an automatic reactor trip from “B” SG low level coincident with a steam flow greater than feed flow mismatch caused by the closure of the “B” MFRV. Closure of the “B” MFRV was the result of a shorted capacitor on the final control card that provides input to the “B” MFRV. The root cause was attributed to organizational and programmatic deficiencies that allowed the card to be placed in service without new upgrades.

LER N2-06-001-00 dated 11/16/06, documents an automatic reactor trip from the “B” SG low level coincident with a steam flow greater than feed flow mismatch caused by closure of the “B” MFRV. Closure of the “B” MFRV was the result of a failed isolator card in the SG water level control system for “B” SG. The isolator card failure was a result of one or more failed transistors in the power supply circuit of the card. The root cause of the transistor failure was age-related degradation.

LER N2-03-001-00 dated 03/31/03, documents an automatic reactor trip from the “C” SG low level coincident with a steam flow greater than feed flow mismatch caused by closure of the “C” MFRV. Closure of the “C” MFRV was the result of a failed driver card in the SG water level control system for “C” SG. The driver card failed as a result of a blown fuse. The corrective actions from this event focused solely on the driver cards. Fuses were inspected on both units with repairs made to several cards.

8.0 ADDITIONAL INFORMATION

Unit 2 continued operating in Mode 1, 100 percent power during this event.